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MONTE CARLO CALCULATIONS OF THE PENETRATION OF NORMALLY INCIDENT NEUTRON BEAMS THROUGH CONCRETE

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ABSTRACT

The O5R Monte Carlo code has been adapted to calculate fast-neutron doses within and outside of concrete slab shields. Such calculations were performed for a variety of shield thicknesses (0 to 180 cm) and for various monoenergetic sources from 0.7 to 14 MeV. The concrete chosen is ordinary concrete with approximately 6% water content. Results are plotted as tissue kerma versus penetration distance for each source energy. The source is in each case a broad normally incident beam.

I. INTRODUCTION

A series of calculations have been performed with an infinite-slab version[1] of the O5R neutron Monte Carlo system[2,3] to determine neutron penetrating doses for various concrete thicknesses and various source energies. There have been many determinations of neutron penetration of

[1] N. A. Betz, Slab Analysis Program, to be published as an ORNL report.

[2] D. C. Irving, R. M. Freestone, F. B. Kam, O5R, A General Purpose Monte Carlo Neutron Transport Code, ORNL-3622 (1965).

[3] F. B. Kam, K. D. Franz, ACTIFK, A General Analysis Code for O5R, ORNL-3856 (1966).

concrete, both experimental and calculated.[4-10] In the face of such a variety and depth of work, we are justified in doing another computation only if it adds something distinctly lacking. The computations reported here have the following unique combination of features:

1. The source energy range from 0.7 to 14 MeV is covered.
2. Each calculation has a monoenergetic source.
3. Inelastic scattering and other neutron-emitting nonelastic processes are treated exactly; that is, according to the best available physical model of the microscopic process. Frequently, such events have been treated as absorptions or as elastic scatterings.
4. The cross sections used are current.[11]

II. CALCULATION PROCEDURE

An infinitely broad, uniform beam of monoenergetic neutrons is allowed to fall normally upon an infinite plane surface bounding a concrete region. The geometry of the concrete region is dichotomous. At every 12 cm (measured normally inward from the bounding surface) two kermas are computed. The

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[4] L. R. Day and M. L. Mullender, Shielding Problems Associated with a Cockroft-Walton Accelerator, AWRE NR-1/63 (1963).

[5] L. N. Zartsev, N. M. Komochkov, and B. S. Sychev, Soviet J. at. Energy, (English Transl.) 12, 558-61 (1963).

[6] F. J. Allen and A. T. Futterer, "Neutron Transmission Data," Nucleonics 21, 120-212 (1963).

[7] K. T. Spinney, Neutron Attenuation in Concrete, AERE T/R 2507 (1958).

[8] D. Spielberg and A. G. Duneer, Dose Attenuation by Soils and Concrete for Broad Parallel Beam Neutron Sources, AN-108 (1958).

[9] E. P. Blizard and J. M. Miller, Radiation Attenuation Characteristics of Structural Concrete, ORNL-2193 (1958).

[10] D. K. Trubey and M. B. Emmett, Some Calculations of the Fast Neutron Distribution in Ordinary Concrete from Point and Plane Isotropic Fission Sources, ORNL RSIC-4 (1965).

[11] D. C. Irving, 05R Cross Section Library, Memo 2, ORNL RSIC Code Package CCC-17 (1965).

first is the kerma that would be measured just outside an infinite plane boundary surface passing through this point and running parallel to the surface of entry. The solid curves in Figs. 1-10 are the plots of these kermas. The second kerma computed at this point is the kerma that would be measured if the concrete region filled the infinite half space bounded by the surface of entry. These kermas are plotted as dashed curves in Figs. 1-10.

Flux densities are estimated at detector surfaces by summing up the surface crossing weights of particle histories divided by the cosine of the angle of crossing. Particles crossing at near grazing angles are not included in the sum. Their contributions are estimated by extrapolation of the smoothed values of the contributions at nongrazing angles.[12] The principal importance sampling scheme employed is the exponential transform.[13,14]

Detailed descriptions of the calculations employed can be found in references 1 and 2.

III. CONCRETE CONSTITUENTS

The calculations were to be performed for ordinary concrete. In shielding parlance ordinary concrete means concrete without very high density additives in the mix. But "ordinary" is not definitive. It varies according to the makeup of the locally available dry constituents, and also according to the amount of water content remaining during the time that the concrete is in use as a shield.

Reference 15 gives the following composition for type 01 ordinary concrete:

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[12] F. H. Clark, "Variance of the Surface Crossing Flux Estimator Used in Monte Carlo Calculations," submitted to Nuclear Science and Engineering.

[13] F. H. Clark, The Exponential Transform as an Importance Sampling Device - A Review, ORNL RSIC-14 (1965).

[14] F. H. Clark and N. A. Betz, Importance Sampling Devices for Selecting Track Lengths and Directions after Scatter in O5R, ORNL TM-1484 (1966).

[15] Reactor Physics Constants, ANL-5800.

Element	Concentration (gm/cm^3)
Hydrogen	0.00484
Oxygen	
in water	0.0384
in dry mix	1.1106
Carbon	0.130
Magnesium	0.00486
Aluminum	0.0119
Silicon	0.438
Sulphur	0.00192
Calcium	0.581
Ferrum	0.00726
Total	2.33

The above concrete description was taken as a starting point. Further study[9,16] indicated that the above water content (under 2% by weight) is extremely low, at least by the standards of the locally available concretes which were analyzed. A water content of 6% by weight, or even somewhat greater, seemed more appropriate.

Further, the precise representation of materials present in only very low concentrations is not necessary for a fast-neutron penetration calculation. (The same statement would be true of a primary gamma-ray penetration problem, but it would not generally be true of a problem involving strong resonance effects, like secondary gamma-ray production, if any of the low concentration materials had large resonances.) Calculation schemes very frequently are limited in the number of different materials they can handle. O5R is no exception to this rule. Consequently, magnesium, aluminum, and sulphur were lumped as quasi-silicon atoms and iron as quasi-calcium. The rule for the lumping was simply this. The atomic density of silicon was adjusted so that the total cross section associated with the adjusted density at 6 MeV was equal to the total 6-MeV cross section of the actual concentration of silicon, magnesium, aluminum, and sulphur given above. Iron was absorbed in calcium in the same manner.

[16] R. E. Maerker, Oak Ridge National Laboratory, private communication.

The revised concrete constituents are as follows:

Element	Concentration [(Atoms/cm^3) x 10^{-24}]
Hydrogen	0.0093
Oxygen	0.0463
Carbon	0.0065
Silicon	0.00992
Calcium	0.00883

With the increased water content the above mixture corresponds to an ordinary concrete of density 2.43 g/cm^3 with a water content of 6% by weight. These were the atomic concentrations used in the calculations.

IV. RESULTS

The calculations were carried out for various monoenergetic source energies and the results are charted by source energy as follows:

Source Energy (MeV)	Fig. No.
14	1
12	2
10	3
8	4
6	5
4	6
3	7
2	8
1.2	9
0.7	10

As previously indicated, the solid curve on each chart gives the kerma value (per source neutron) per cm^2 behind a slab of thickness indicated on the abscissa. The dashed curve gives the corresponding kerma for penetrating the abscissa distance into an infinite half space of concrete. The ratio of such corresponding values can be used to estimate boundary correction effects. Such estimates may then be used, crudely, to estimate finite region effects from other infinite region calculations.[10]

We do not have available a good set of data for direct comparisons. However, for quite similar concretes and a fission spectrum source, Blizard and Miller[9] found an asymptotic dose relaxation length of 11.5 cm. The same workers, using removal cross sections for the elemental constituents of their concrete,[17] estimate a dose relaxation length of 11.3 cm.

The results given in this report are for monoenergetic sources and therefore are not directly comparable with fission source measurements. The most penetrating source energy we deal with, however, is 6 MeV for which the asymptotic relaxation length, determined from Fig. 5, is 11.5 cm. The two next most penetrating source energies are 8 and 10 MeV and their relaxation lengths, determined from Figs. 3 and 4, are both 11.3 cm.

It is this energy region that ordinarily establishes penetrating characteristics of a thick shield for fission spectrum sources. The correspondence of results must therefore be taken as substantial validation.

The statistical uncertainties (relative standard errors) in the results vary generally as the amount of the attenuation. Most of the computed attenuations of factors up to 10^{-3} had statistical errors less than 5%, a few somewhat greater. Errors in attenuations to 10^{-4} ranged up to 15% and those in attenuations to 10^{-5} ranged up to 25%. For greater attenuations most errors were in the range of 25-30%, with a few as high as 50%. The fitting of curves to points with statistical uncertainties is, however, a stabilizing factor of great power, but one whose magnitude cannot be precisely assessed. A point read from such a curve can be expected to have an inherent accuracy much greater than the statistical accuracy associated with the calculation of a single nearby point.

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[17] G. T. Chapman and C. L. Storrs, Effective Neutron Removal Cross Sections for Shielding, ORNL-1843 (1955).

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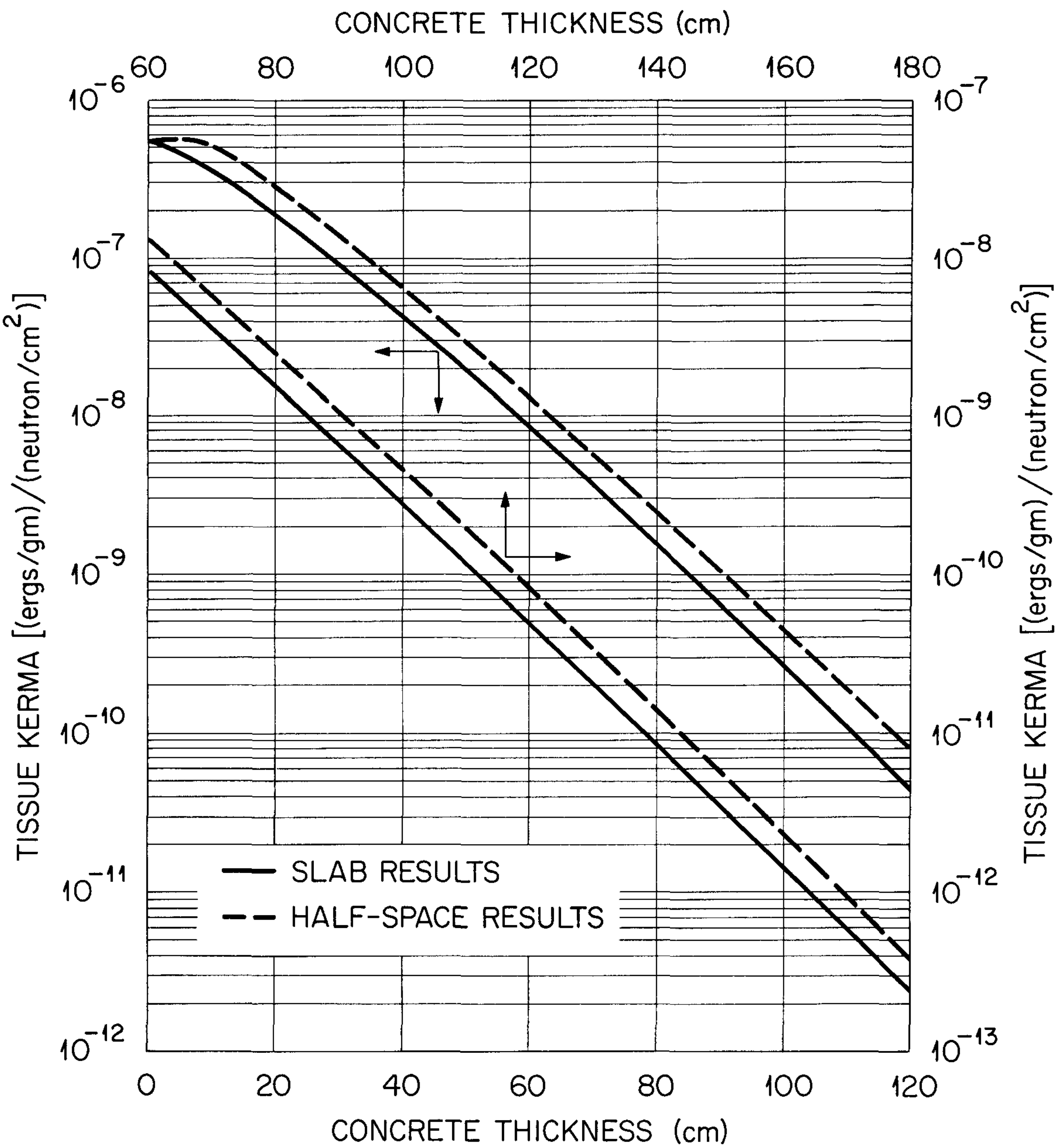

Fig. 1. Penetration of Normally Incident Neutrons of Energy 14 MeV Through Slabs and into Half Spaces of Ordinary Concrete.

ORNL-DWG 66-6442

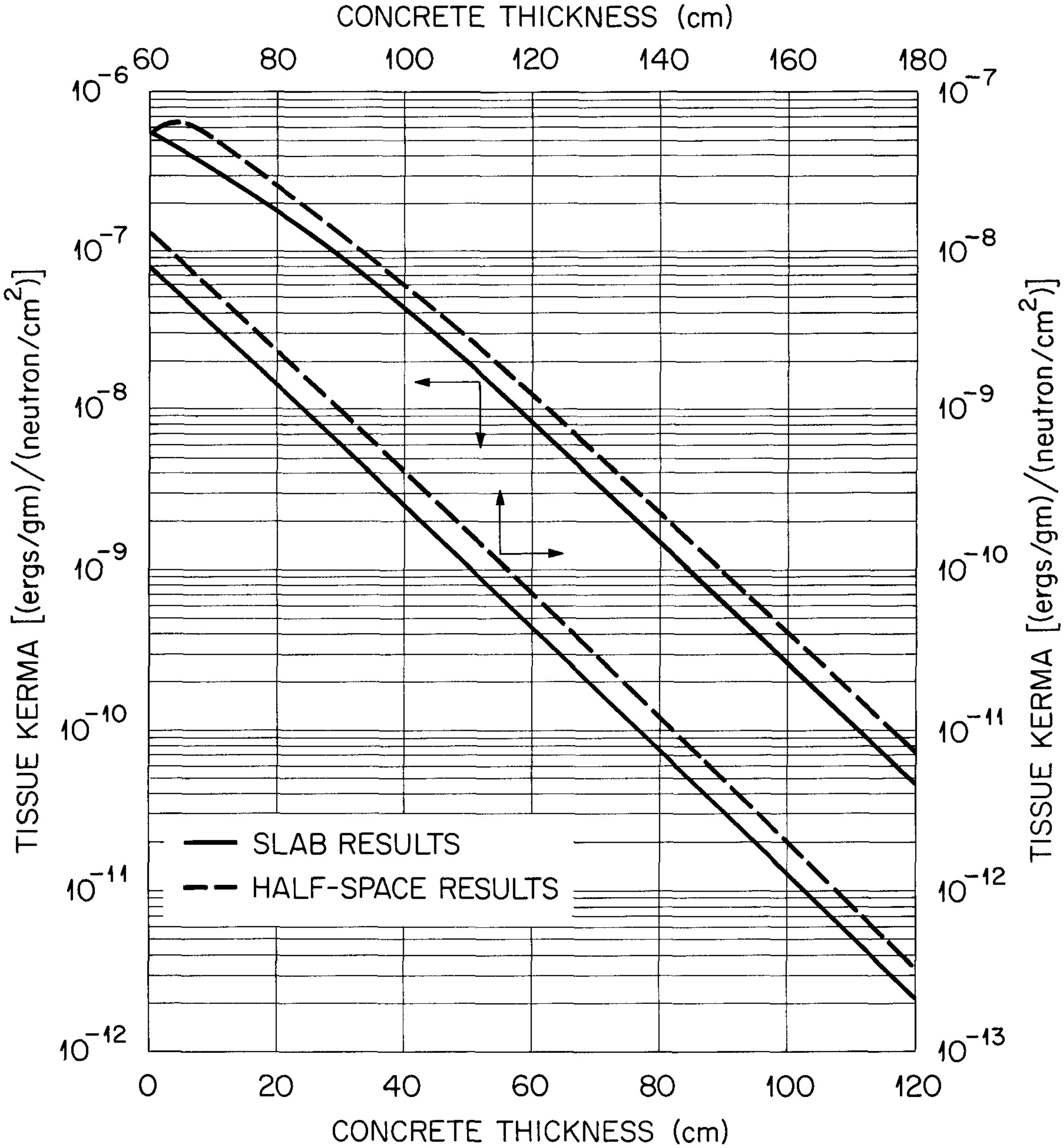

Fig. 2. Penetration of Normally Incident Neutrons of Energy 12 MeV Through Slabs and into Half Spaces of Ordinary Concrete.

ORNL-DWG 66-6443

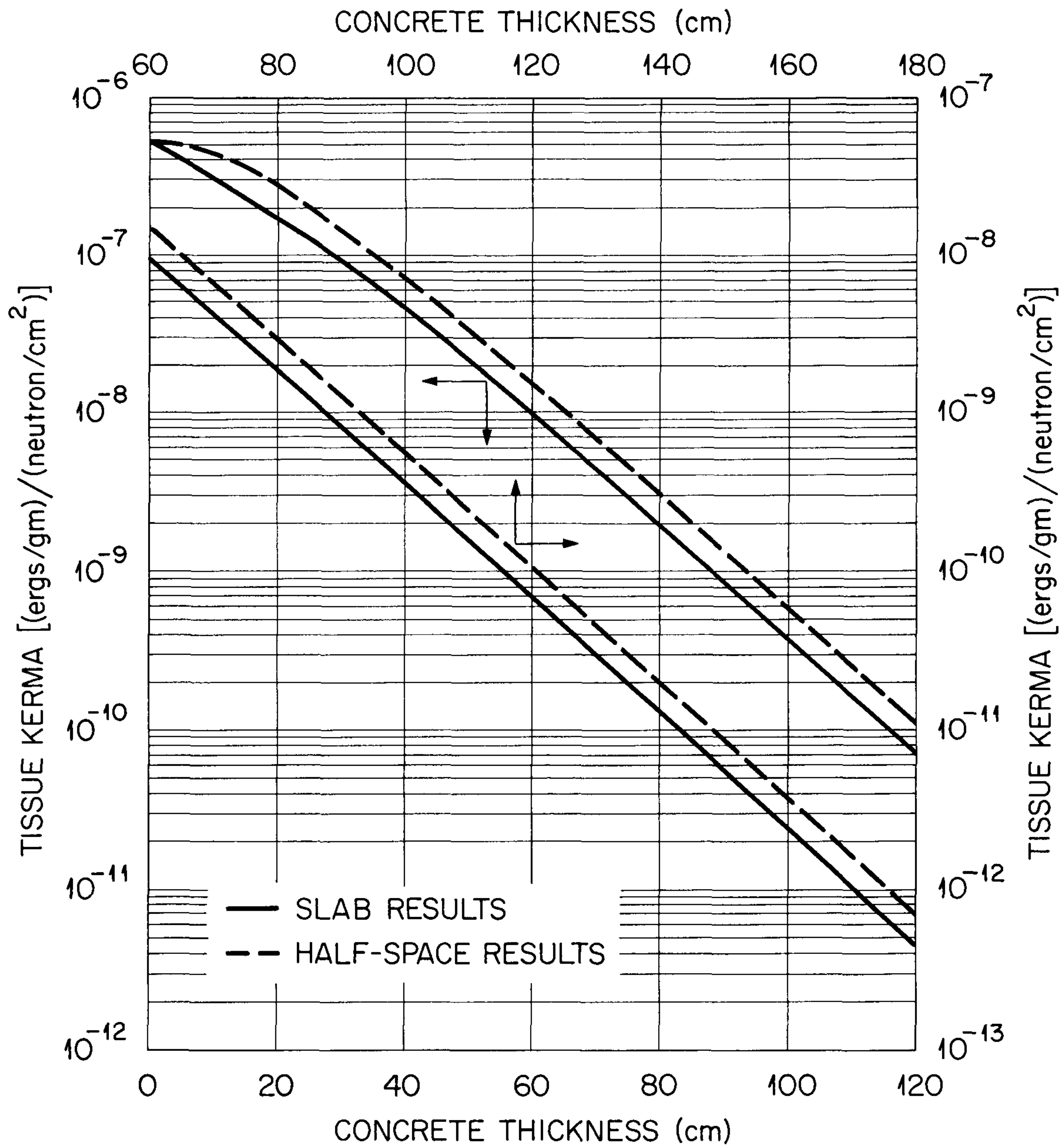

Fig. 3. Penetration of Normally Incident Neutrons of Energy 10 MeV Through Slabs and into Half Spaces of Ordinary Concrete.

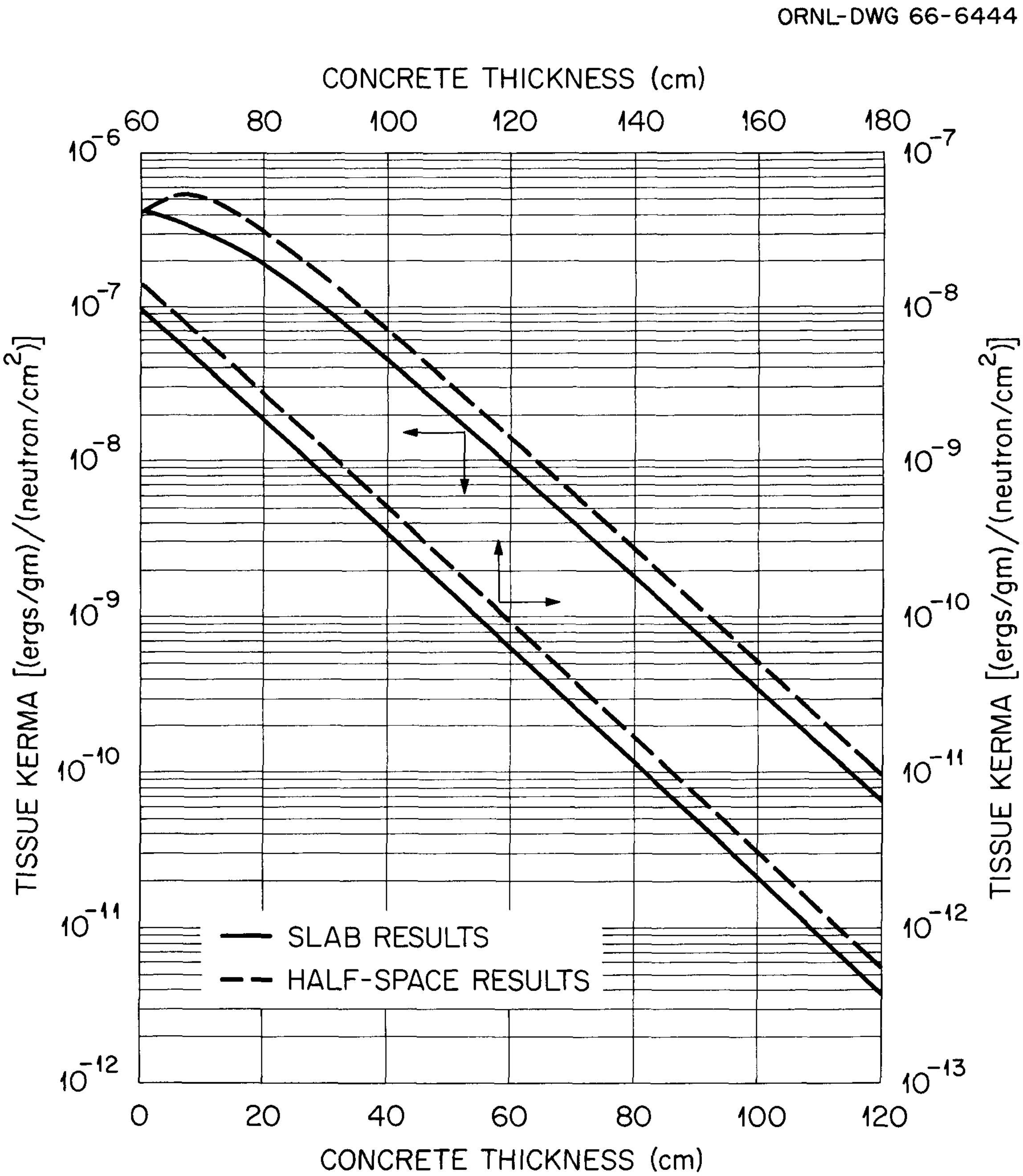

Fig. 4. Penetration of Normally Incident Neutrons of Energy 8 MeV Through Slabs and into Half Spaces of Ordinary Concrete.

ORNL-DWG 66-6445

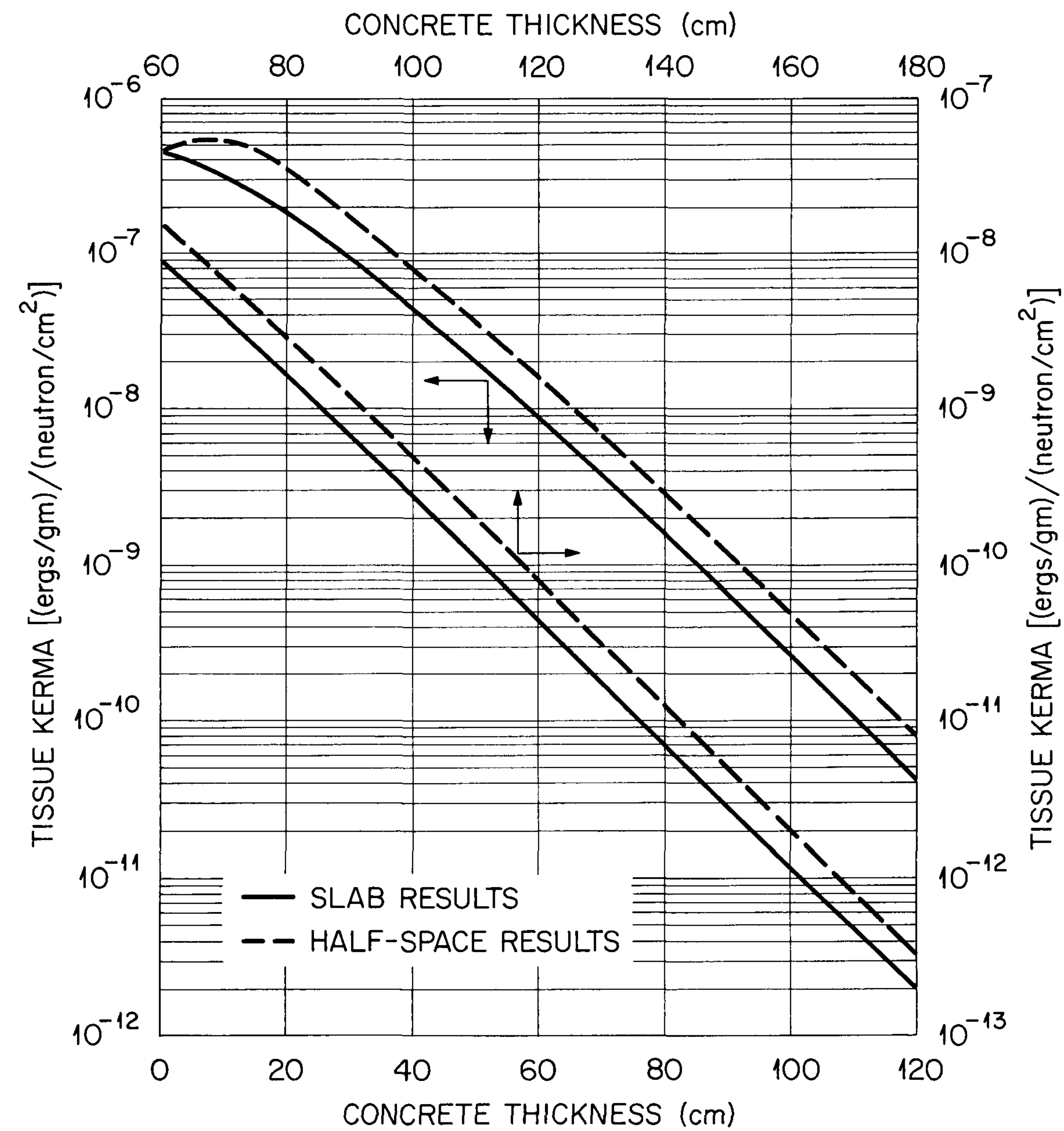

Fig. 5. Penetration of Normally Incident Neutrons of Energy 6 MeV Through Slabs and into Half Spaces of Ordinary Concrete.

ORNL-DWG 66-6446

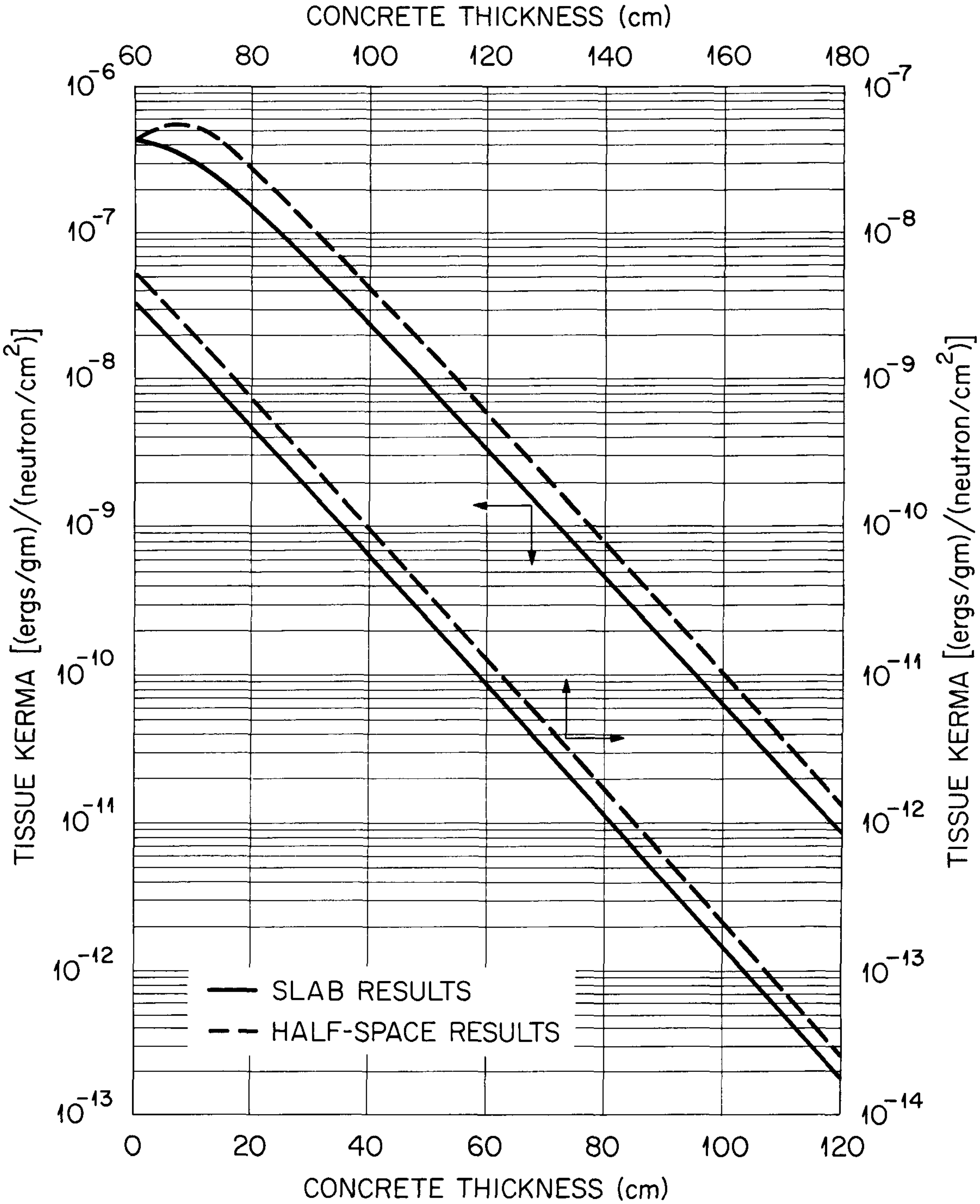

Fig. 6. Penetration of Normally Incident Neutrons of Energy 4 MeV Through Slabs and into Half Spaces of Ordinary Concrete.

ORNL-DWG 66-6447

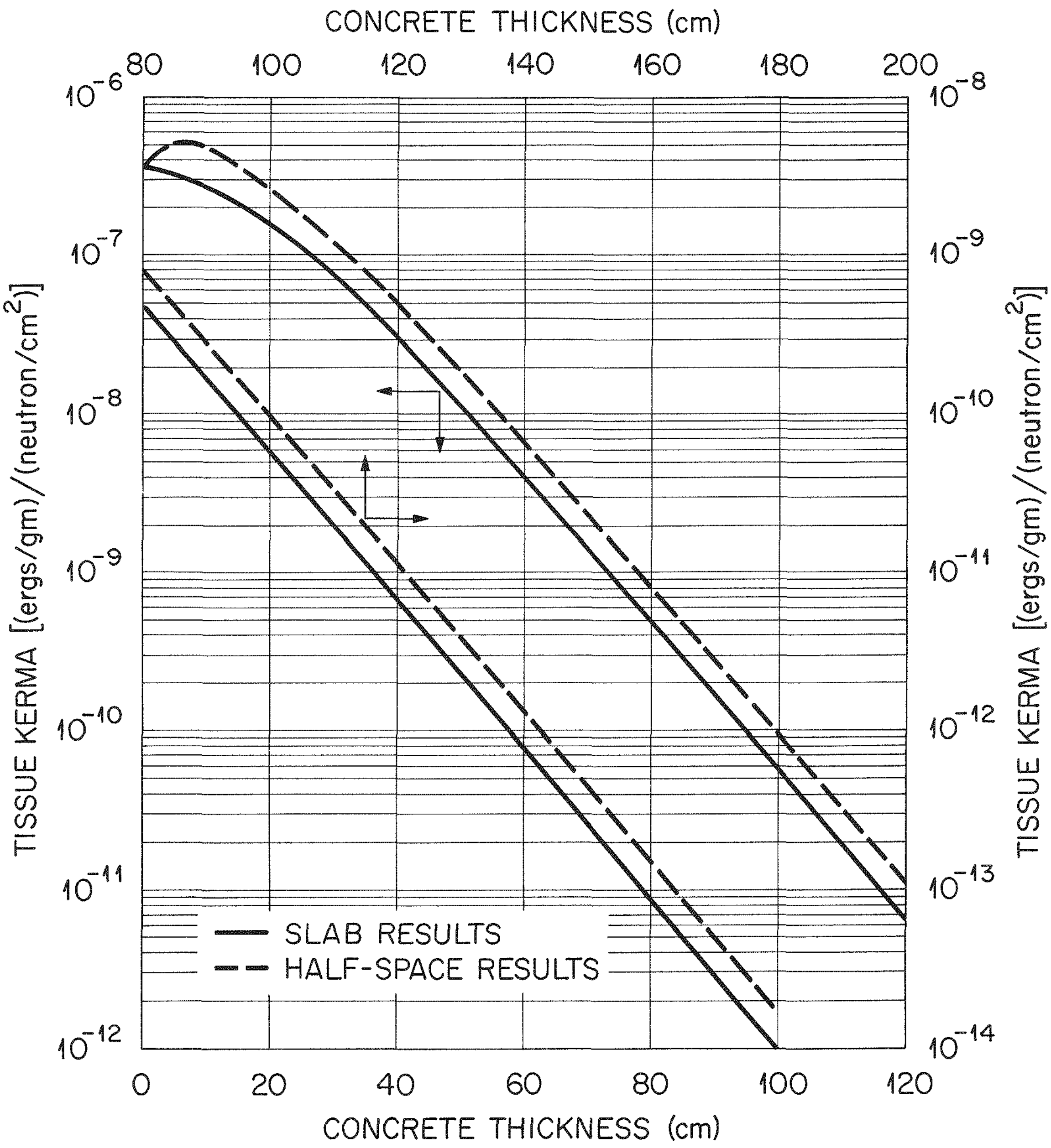

Fig. 7. Penetration of Normally Incident Neutrons of Energy 3 MeV Through Slabs and into Half Spaces of Ordinary Concrete.

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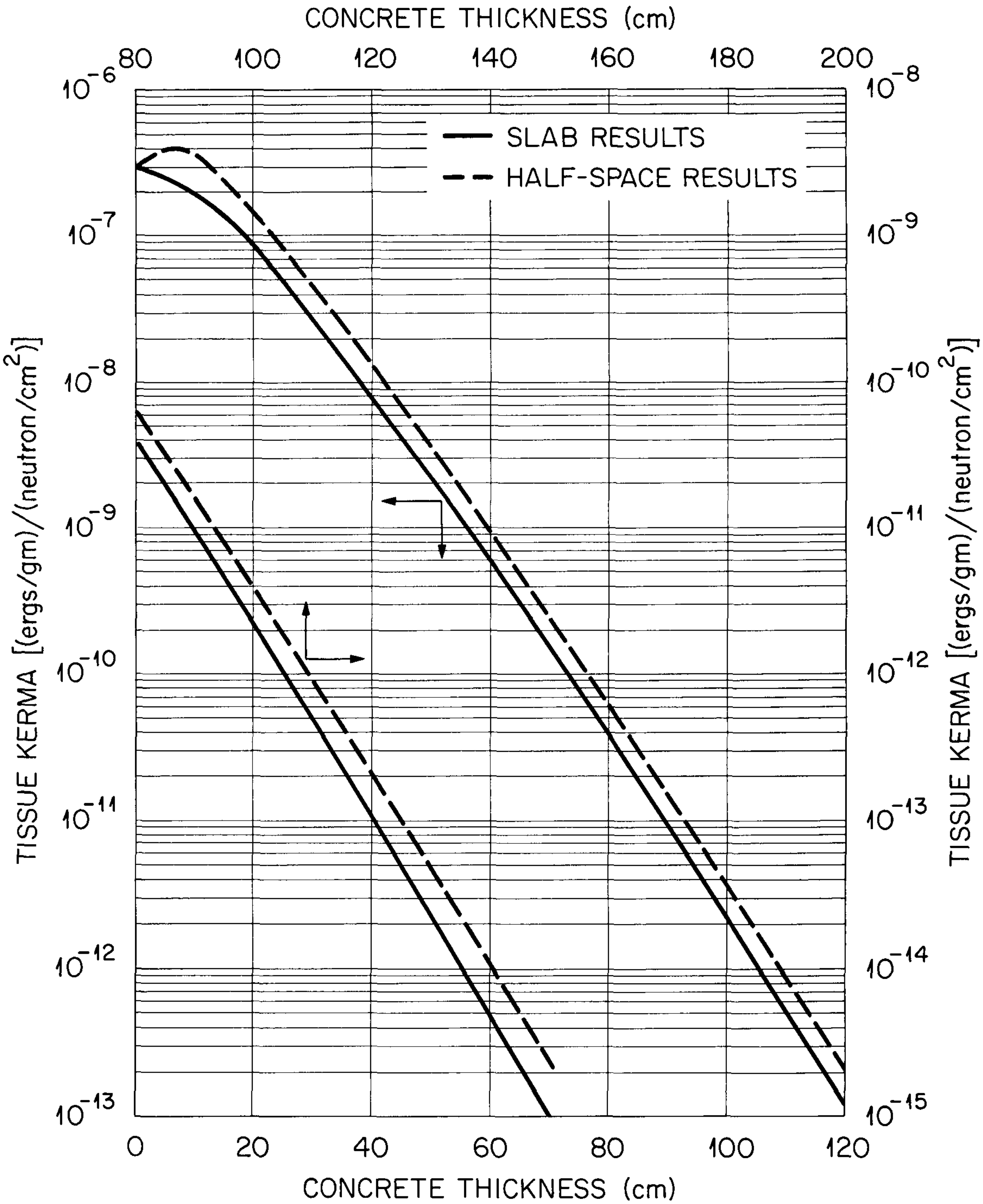

Fig. 8. Penetration of Normally Incident Neutrons of Energy 2 MeV Through Slabs and into Half Spaces of Ordinary Concrete.

ORNL-DWG 66-6449

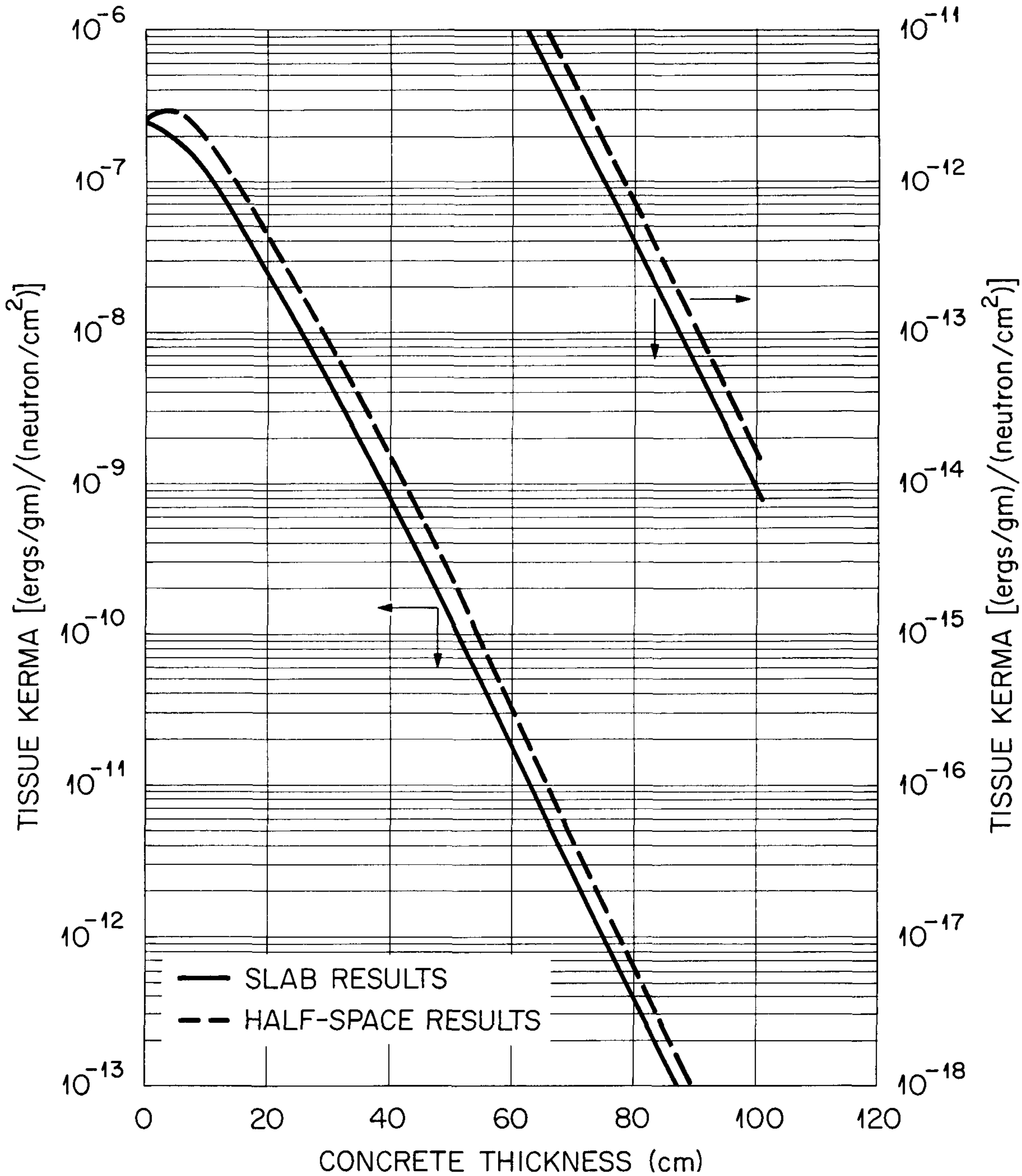

Fig. 9. Penetration of Normally Incident Neutrons of Energy 1.3 MeV Through Slabs and into Half Spaces of Ordinary Concrete.

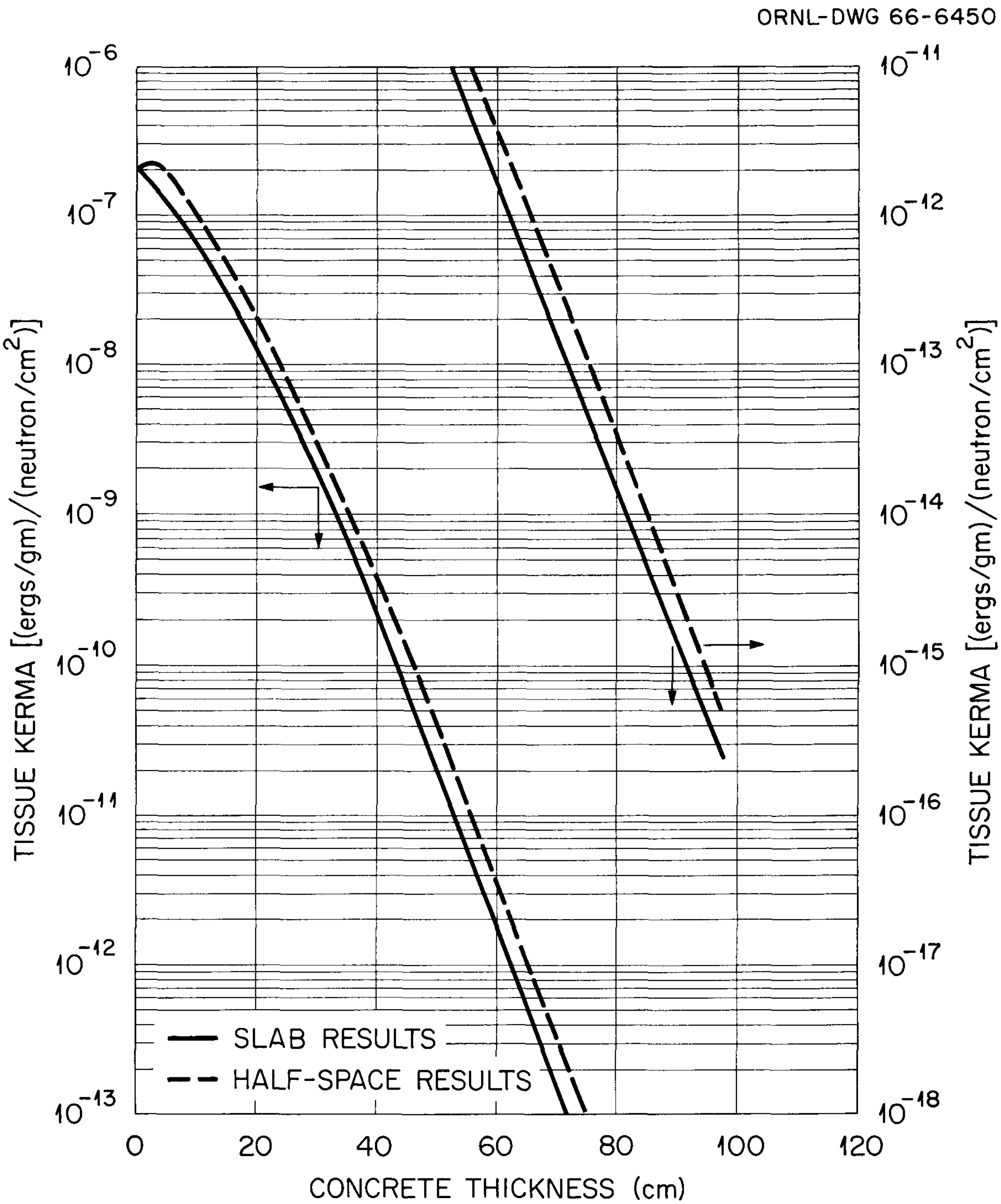

Fig. 10. Penetration of Normally Incident Neutrons of Energy 0.7 MeV Through Slabs and into Half Spaces of Ordinary Concrete.

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